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March 31, 2003

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: LER –2003-001-00 Manual Reactor Scram (#106) Due To Loss Of Condensate/Feedwater]

Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

GNRO-2003/00015

Ladies & Gentlemen:

Attached is Licensee Event Report (LER) 2003-001-00 which is a final report.
This letter does not contain any commitments.

Yours truly,

JCR/JEO:jeo
attachments: 1 LER 2003-001-00
cc: (See Next Page)

cc:

Hoeg	T. L.	(GGNS Senior Resident)	(w/a)
Levanway	D. E.	(Wise Carter)	(w/a)
Reynolds	N. S.		(w/a)
Smith	L. J.	(Wise Carter)	(w/a)
Thomas	H. L.		(w/o)

U.S. Nuclear Regulatory Commission ATTN: Mr. E. W. Merschoff (w/2) 611 Ryan Plaza Drive, Suite 400 Arlington, TX 76011-4005
U.S. Nuclear Regulatory Commission ATTN: Mr. David H. Jaffe NRR/DLPM (w/2) **ATTN: FOR ADDRESSEE ONLY** ATTN: U.S. Postal Delivery Address Only Mail Stop OWFN/7D-1 Washington, D.C. 20555-0001

ATTACHMENT 1 TO GNRO-2003/00015
LICENSEE-IDENTIFIED COMMITMENTS

Letter #:	GNRO-2003/00015

COMMITMENT	TYPE (Check only one type)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
N/A			
N/A			
N/A			
N/A			

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Grand Gulf Nuclear Station, Unit 1	05000 416	1 OF 6

4. TITLE

Manual Reactor SCRAM Due To Loss Of Condensate/Feedwater

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER 05000
01	30	2003	2003	001	00	03	31	2003	FACILITY NAME N/A	DOCKET NUMBER 05000

9. OPERATING MODE	10. POWER LEVEL
1	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	X	OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
James E. Owens, Senior Licensing Specialist	601-437-6219

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On January 30, 2003, at approximately 10:54 a.m., Grand Gulf Nuclear Station was operating at 100 percent power when operators inserted a manual reactor scram. The manual scram was initiated due to decreasing reactor water level resulting from a loss of feedwater. The loss of feedwater was caused by isolation of the condensate demineralizers due to shorting of a conductor which tripped the common power supply for the position indication limit switches for the demineralizers valves. The loss of the inlet valve's position indication caused all operating demineralizer outlet valves to close.

High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) auto initiated and injected into the reactor vessel to restore level.

In response to the HPCS initiation the Division 3 Emergency Diesel Generator auto started but was not needed to supply the Division 3 Buss.

As designed, Standby Service Water system loops "A" and "C" started, for required support of equipment.

A. REPORTABLE OCCURRENCE

Licensee Event Report (LER) Reportable Events:

1. Manual reactor scram from steady state condition resulting from decreasing reactor pressure vessel water level due to loss of feedwater 10CFR50.73(a)(2)(iv).
2. High Pressure Core Spray and Reactor Core Isolation Cooling auto initiation 10CFR50.73(a)(2)(iv).
3. Valid auto start of the Division 3 Emergency Diesel Generator 10CFR50.73(a)(2)(iv).
4. Initiation of Standby Service Water "A" and "C" (Ultimate Heat Sink) 10CFR50.73(a)(2)(iv).
5. General Containment Isolation Signal 10CFR50.73(a)(2)(iv).
6. HPCS Injection into the reactor vessel is reportable as a special report per Grand Gulf Technical Requirements Manual section 7.7.2.1.

Events Not Reportable But Mentioned in This LER:

1. Exceeded 100 degree per hour heat up/cool down rate.

On January 30, 2003, at 10:54 a.m. with the reactor mode switch in run and the reactor operating at 100 percent power, Grand Gulf Nuclear Station (GGNS) operators initiated a manual reactor scram due to decreasing reactor water level caused by a loss of feedwater. The manual scram is reportable pursuant to 10 CFR 50.73(a)(2)(iv).

The reactor trip was the result of isolation of all operating condensate demineralizers [SF] due to a single electrical short. Isolation of the condensate demineralizers led to a trip of all three condensate booster pumps and both turbine driven feed pumps which resulted in loss of feedwater [SJ] to the reactor vessel. As reactor water level was being recovered by the High Pressure Core Spray system (HPCS) [BG] and the Reactor Core Isolation Cooling system (RCIC) [BN], a condensate booster pump and a turbine driven feed pump were restarted by operators from the control room. Feedwater was recovered without extraordinary effort. Therefore this event did not constitute a loss of heat sink.

As a result of the loss of feedwater and corresponding decreasing reactor pressure vessel (RPV) water level the High Pressure Core Spray and RCIC systems auto initiated on Level 2 (-41.6 inches) low RPV water level auto initiation set point. Upon reaching the level 2 set point, a Containment Isolation [JM] signal was initiated and the Standby Service Water (SSW) [BI] system "A" auto started in support of the RCIC system and the "C" loop of SSW started in support of HPCS. Initiation of these systems is reportable per 10CFR50.73(a)(2)(iv). The HPCS injection is included in this LER as a Special Report per GGNS Technical Requirements Manual section 7.7.2.1.

The Division 3 Emergency Diesel Generator (EDG) [EK] received an auto start signal as a result of the Level 2 setpoint. The EDG started but did not synchronize to the Division 3 Buss because the normal feed to the buss was never lost. The auto start of the Division 3 EDG is reportable pursuant to 10 CFR 50.73(a)(2)(iv).

During the on-shift post trip analysis performed following recovery from the scram, it was discovered that Technical Specifications (TS) 3.4.11 which limits heat up/cool down to no more than 100 degrees Fahrenheit per hour was exceeded. This was based on the reactor bottom head drain temperature recorded on the control room data sheet. In accordance with TS 3.4.11 an analysis was performed to determine reactor coolant system (RCS) acceptability for restart. The analysis determined that even though heat up/cool down rate exceeded 100 degrees Fahrenheit per hour, this was bounded by the original analyses and therefore the RCS was acceptable for restart. Later during performance of the more in-depth off shift post trip analyses it was discovered that the heat up/cool down rate for one of the recirculation loops exceeded 100 degrees per hour and that heat up/ cool down rates were higher than those noted on the control room data sheet. It was also noted that there was a difference between the Plant Display System (PDS) computer point temperature reading and Vessel Temperature Chart recorder. Analyses per TS 3.4.11 were performed utilizing data from both the PDS computer point and the Vessel Temperature Chart recorder. The result indicated that all heat up/cool down rates were bounded by the original analysis of this event and therefore the RCS was acceptable for restart.

There were no safety system functional failures associated with this event.

B. INITIAL CONDITIONS

At the time of the event, the reactor was in OPERATIONAL MODE 1 with reactor power at approximately 100 percent and reactor pressure at approximately 1030 psig. Moderator temperature was approximately 540 degrees Fahrenheit and reactor level was approximately 36 inches.

C. DESCRIPTION OF OCCURRENCE

On January 30, 2003, with the plant at 100 percent power, with the "G" condensate demineralizer out of service, Mechanical Maintenance had just completed an air operated valve (AOV) actuator rebuild for 1N22-F045G ("G" condensate demineralizer inlet valve). Instrument and Control (I&C) technicians were in the process of re-mounting the limit switch assembly onto the valve actuator when a manual reactor scram was initiated.

A nut connecting flexible conduit to a 45 degree elbow was loosened and disconnected. This allowed the limit switch assembly for the 1N22-F045G to be maneuvered around interferences in order to align the mounting holes. The wiring had not been disconnected from the limit switch. It is established practice to work the limit switch mounting energized.

The technicians had reinserted the flex conduit into the 45 degree elbow fitting and were tightening the flexible conduit nut into place. As the technician neared completion of hand tightening the nut to the flexible conduit, they heard unusual flow noises and piping vibrations. They secured the job and exited the area.

Investigation revealed that as they seated the flexible conduit into the 45 degree elbow, an apparent pre-existing nick in the insulation failed allowing a bare wire to make contact with the grounded elbow. This caused a breaker (2L1) in the Condensate Cleanup Control Panel (1H22-P474) to trip. The breaker trip de-energized position indication for the demineralizer inlet valves. The loss of indication on the inlet valves caused the outlet valves for all in-service demineralizers to go closed.

The demineralizer outlet valves are interlocked to the inlet valves such that when the inlet valves are not full open, the outlet valves will automatically close. The other valves in the system have their indication powered directly from the 2L1 source and therefore lost all associated indication at Panel 1H22-P474.

The reference drawings show that when the limit switch circuits for the demineralizer inlet valves lost power, associated control relays de-energized which resulted in two effects:

- All demineralizer inlet valves indicated dual position
- The respective position indicating contacts for all demineralizer inlet valves opened, removing the control signal to all demineralizer outlet valves' current-to-pneumatic (I/P) controllers, simultaneously closing all 7 operating demineralizer outlet valves.

D. APPARENT CAUSE

Historical design of the power supplies within the 1H22-P474 panel does not meet current design engineering practices to isolate failures to individual trains. The system is designed with a molded case breaker as a common power supply for the valve position limit switches for the eight demineralizers and the three resin regeneration tanks. There is no downstream fault isolation for an individual demineralizer or resin regeneration tank valve position. The circuit breaker supplying power to this equipment is the only fault protection.

There is no licensing or electrical design standard that requires this balance of plant system to be resistant to single active failures.

Additionally, a change in the control system design by Startup Engineers (SFR-1-E-1875) introduced the interlock between the inlet and outlet valves that was not part of the original design. The interlock was intended to prevent inadvertent over-pressurization of the cation tank if a condensate demineralizer outlet valve was left partially or completely open. This interlock created an unintended single failure vulnerability into the controls in the event that the limit switch supply breaker should trip.

Another key factor was the configuration of the limit switch mounting plate. Attachment of each individual flex conduit encourages flow vibration to cause movement of the limit switch wires within the flex conduit and connecting elbow. The resulting movement caused the rough inside surface of the conduit elbow to rub the insulation of the wire until the insulation failed. This root cause is strongly supported by the metallurgical evaluation of the cast elbow and the wire from the limit switch failure in the feedwater demineralization system.

E. CORRECTIVE ACTIONS

Immediate Actions

The following corrective actions were taken immediately following the event.

Access to the Demineralizer and Resin Regeneration Rooms was temporarily limited and "High Impact Areas" postings were hung. Readily accessible 1N22-F045 valve limit switches were inspected for wiring damage and Bailey Power Supplies have been added to the Power Supply Program.

Long Term Actions:

Condition Report CR-GGN-2003-0300 was written

F. SAFETY ASSESSMENT

All Emergency Core Cooling Systems and the RCIC system were available to provide makeup following the loss of feedwater. All safety systems performed as designed.

The core damage risk associated with this event was evaluated with Grand Gulf's online risk monitor (EOOS) in order to characterize its safety significance. The EOOS model that was used is based on the most recent update of the Grand Gulf Probabilistic Risk Assessment. This event was simulated by removing the condensate and feedwater systems from service and setting the Loss of Feedwater initiator frequency to a value of 1.0. In addition, the frequencies for other transient initiators (Loss of Offsite Power, Loss of Power Conversion System (PCS), PCS Available, Loss of Turbine Building Cooling Water, Loss of Component Cooling Water, Loss of Instrument Air and Loss of Plant Service Water) were set to zero. Since no other modeled systems were out of service, no other changes were necessary. This model was then quantified at a truncation level of 1E-09/yr. The resulting core damage frequency (CDF) was 2.21E-06/yr which is an increase of 3E-08/yr above the no maintenance baseline model CDF of 2.18E-06/yr. This calculated increase in CDF is not considered to be risk significant since it is well below the Regulatory Guide 1.174 guidance for a very small increase in CDF (less than 1E-06/yr).

G. ADDITIONAL INFORMATION

HPCS injected at a flow rate of approximately 4600 gallons per minute. The temperature of the source water was approximately 90 degrees Fahrenheit. The vessel pressure was approximately 955 psig at the time of the injection. This is the fifteenth cycle of the HPCS system experienced at GGNS at rated temperature and pressure. The current value of the nozzle usage factor is still within 0.70. Report of the ECCS injection is being submitted as part of this LER in accordance with the Special Reporting requirements of GGNS Technical Requirements Manual section 7.7.2.1

Pursuant to 10CFR50.73(b)(5) the licensee considered this event to be a frequent event. There has not been any occurrence of the same underlying concern in the past two years at Grand Gulf Nuclear Station.

Additionally, a modification to the condensate system to remove the interlock between the demineralizer inlet and outlet valves has been completed.

Note:

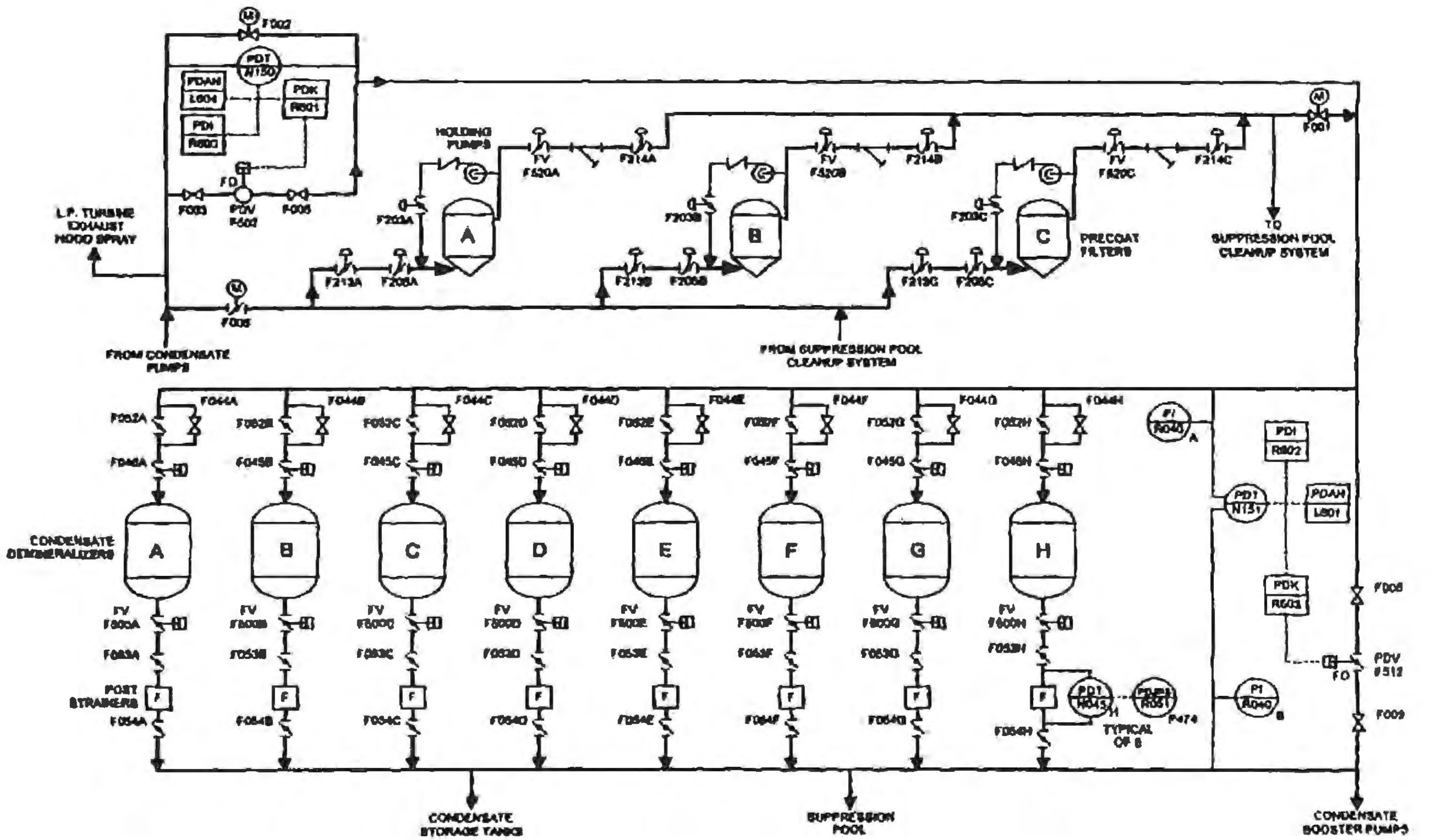
Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ]

Attachments:

Schematic of the condensate demineralizer system.

ATTACHMENT TO LER 2003-001

GG-1-FIG-OP-N2200



Condensate Cleanup System
Figure 1